NASA TT F-10,155

USE OF THE METHOD OF "REMOVAL CROSS SECTION" FOR THE COMPUTATION OF SHIELDING NOT CONTAINING HYDROGEN

D.L.Broder, A.A.Kutuzov, V.V.Levin, and V.V.Frolov

Translation of "Primeneniye metodiki 'secheniy vyvedeniya' dlya rascheta zashchity, ne soderzhashchey vodorod". Fiziki Zashchity Reaktorov (Problems of Reactor Shielding Physics), Gosatomizdat, Moscow, 1963. P. 52-60.

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION
WASHINGTON
MAY 1966

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The method of removal cross section, equivalent to the neutron absorption cross section in the group with an energy interval of $\Delta E = E - E_{thr}$ (where E_{thr} is the threshold energy), is used for computing hydrogen-free shieldings in which media other than water (e.g., boron carbide or aluminum) are used as moderator. Diagrams and Tables of the spatial distribution of neutrons with energies of 4 and 14.9 Mev in Th^{232} and U^{235} fission chambers show that the removal cross section reaches saturation at relatively short distances r_0 from source to detector and that at distances greater than r_0 the cross section, in a homogeneous mixture, is close to that measured in heterogeneous types. The method is applicable also to heavy moderators such as iron, although considerable underestimates occur because of the dependence of the removal cross section on the atomic weight of the moderator.

author

The method of the removal cross section is a highly convenient method of calculating the spatial distribution of neutron fluxes in hydrogen-containing shielding (Bibl.1, 2).

This method of calculating shieldings, which is relatively simple, consists essentially in finding that cross section of an element entering into the shield-

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* Numbers in the margin indicate pagination in the original foreign text.

ing composition which is equivalent to the absorption cross section for neutrons in the group with an energy range of $\Delta E = E - E_{thr}$. This cross section, which is known as "removal cross section", is composed of the absorption cross section, part of the elastic scattering cross section, and most of the inelastic scattering cross section.

Elastic scattering by light nuclei affects the absorption of fast neutrons (their removal from the energy interval $E - E_{thr}$), both as a result of the loss of energy by the neutrons during scattering and as a result of lengthening of the neutron path in the shielding, owing to the changed direction of travel. Only the latter is significant for heavy nuclei. Inelastic processes lead to a greater energy loss with lesser anisotropy of neutron scattering and thus make a greater contribution to the removal cross section at a rather high initial neutron energy. The fast-neutron flux, for a source emitting a neutron spectrum, is calculated by this method, using the formula /53

$$\Phi(z) = \int_{E_{thr}}^{\infty} S(E) D(E, E_{thr}, \theta z) G(z) e^{-\Sigma_{rem}(E)(1-\theta)z} dE,$$

where $S(E)$ is the neutron spectrum of the source; Σ_{rem} is the removal cross section; $D(E, E_{thr}, \theta_z)$ is the spatial distribution of fast neutrons in the energy interval $E - E_{thr}$ at the initial energy E; $G(z)$ is a geometric factor; z is the distance from the source to the point of measurement; θ is the volume concentration of the hydrogen-containing moderator.

It must be noted that the applicability of this method presupposes the satisfaction of certain conditions:

1. The distribution of the neutron flux is calculated in a medium consisting of a hydrogen-containing moderator, in which the substance moderating the neutrons over the removal cross section is homogeneously distributed or is

heterogeneous and occurs in the form of blocks.

2. A layer of this moderator, located between the block of "removing" substance and the detector, or (in the case of a homogeneous mixture) the distance from the source to the detector, must not be less than a certain value r_0 (E, E_{thr}). In addition, r_0 depends on the properties of the "removing" substance. This method can also be used at smaller thicknesses, but then the value of the removal cross section must be decreased.

A direct application of the removal cross section method to a calculation of all groups of moderating neutrons is difficult in some cases, for example in homogeneous reactors, since the value of the removal cross section, measured by the aid of a detector sensitive to slow neutrons, may vary with the composition of the homogeneous mixture. At the same time, this effect is considerably less for fast neutrons.

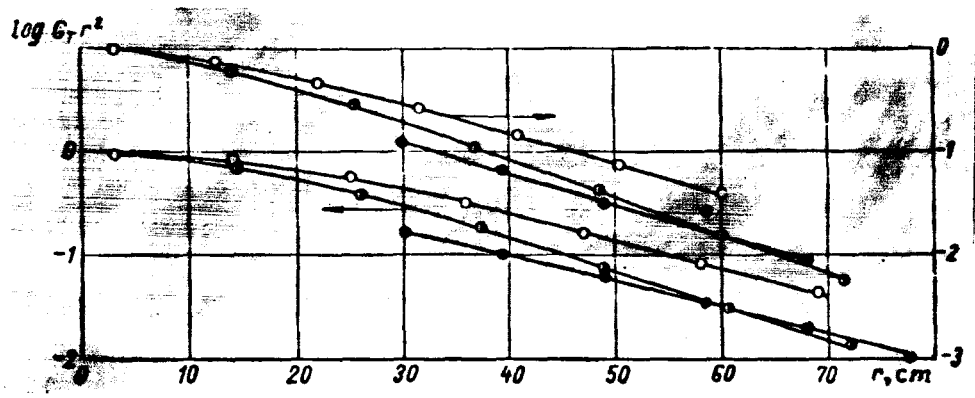

Fig.1 Spatial Distribution of Fast Neutrons Measured in a Th^{232} Fission Chamber (for the Left Ordinate, E = 14.9 Mev; for the Right Ordinate, E = 4 Mev)
o - In aluminum; ● - In a homogeneous mixture of iron with aluminum (thickness of iron layer 2 cm, aluminum 9.5 cm); ● - In aluminum behind an iron moderator block of 10 cm thickness.

The fluxes of neutrons formed as a result of the moderation of fast neutrons were assumed to have been calculated by age dating (Bibl.3, 4). The moderator sources used were spatial distributions of the group of fast neutrons

calculated by the method of removal cross sections.

It has been postulated (Bibl.4, 5) that if the question of the accumulation of low-energy neutrons is disregarded, one could attempt to apply the removal cross section method to a medium containing light moderators other than hydrogen. Those authors investigated the use of boron carbide mixed with iron or lead as moderator. The removal cross sections of iron, lead, and water in homogeneous mixture with boron carbide, or placed in it in slug or block form, were measured. It was found that the removal cross sections of these substances in boron carbide, measured in fission chambers with Th^{232} and U^{235}, coincide and are close in value to the removal cross sections of these substances measured in water.

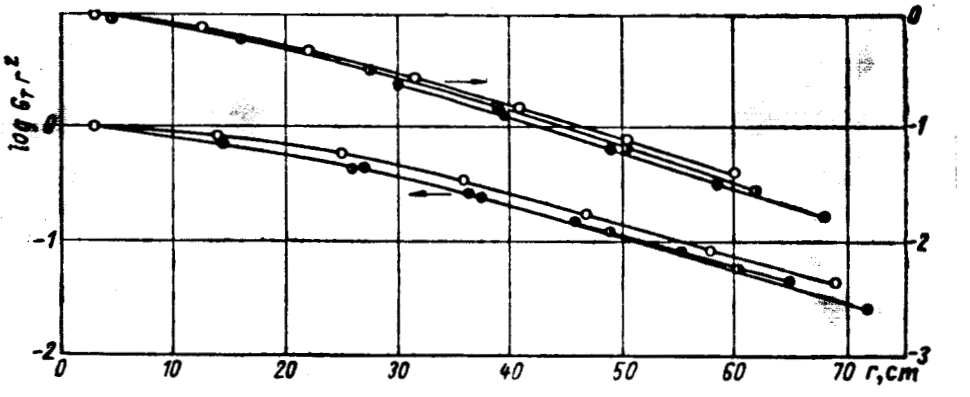

Fig.2 Spatial Distribution of Fast Neutrons Measured in a Th^{232} Fission Chamber (for Left Ordinate, E = 14.9 Mev; for Right Ordinate, E = 4 Mev)
o - In aluminum; ● - In homogeneous mixture of Pb and Al (thickness of lead layers, 2 cm; thickness of aluminum layers, 9.5 cm);
● - In aluminum behind a lead moderator block of 10 cm thickness.

In this paper, we present additional experimental data proving the possibility of generalizing the removal cross section method and applying it to heavier moderators. As moderator, we used aluminum in this case. The neutron sources were of E = 4 Mev and 14.9 Mev energy. We also measured the removal cross sections of iron and lead in boron carbide on the fission neutron spectrum

and the removal cross section of iron on the spectrum of the PW reactor* (Bibl.6).

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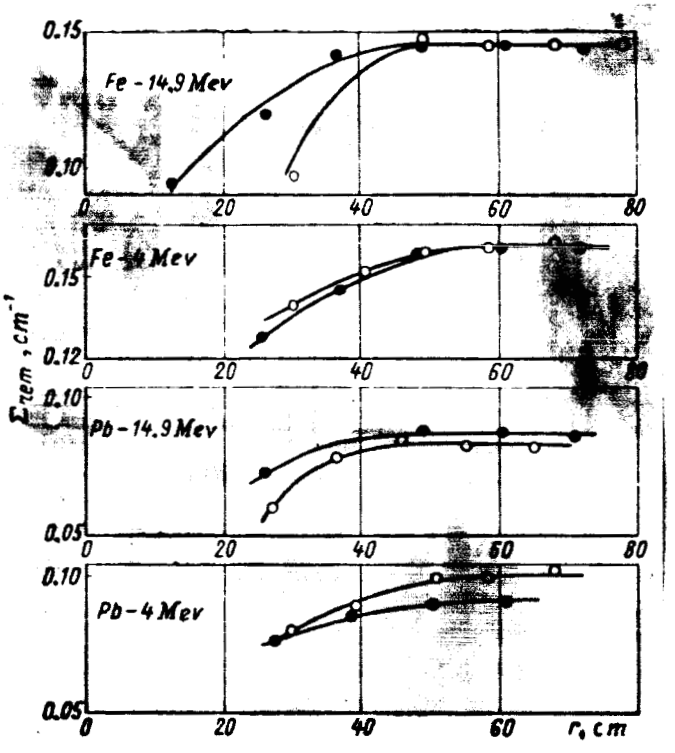

Fig.3 Relation between Removal Cross Sections of Iron and Lead in Aluminum for Neutrons of Energies E = 4 Mev and 14.9 Mev
● - Distance between source and detector in homogeneous mixtures;
o - Distance between block of removing material and detector.

Experimental Setup /55

The neutron sources used were the reactions D (d, n) He^3 with an initial neutron energy E = 4 Mev, T (d, n) He^4 (E = 14.9 Mev), and a disk of U^{235} exposed to a beam of fast neutrons; the uranium had been obtained from the reactor of the world's first atomic power station.

The sources were in the form of disks, 10 mm in diameter for the monoenergetic neutrons, and 46 mm in diameter for the fission spectrum.

* PW reactor = PWR = water-cooled, water-moderated (pressurized water) reactor.

Detection of the fast neutrons was done in Th^{232} fission chambers. A detailed description of the detectors is given elsewhere (Bibl.4, 5). The geometry of the experiment was similar to that described earlier (Bibl.5). Aluminum prisms were assembled from 1000 × 1000 × 10 mm sheets. The iron and lead plates were 710 × 710 × 10 mm, from which blocks of thicknesses up to 100 mm were

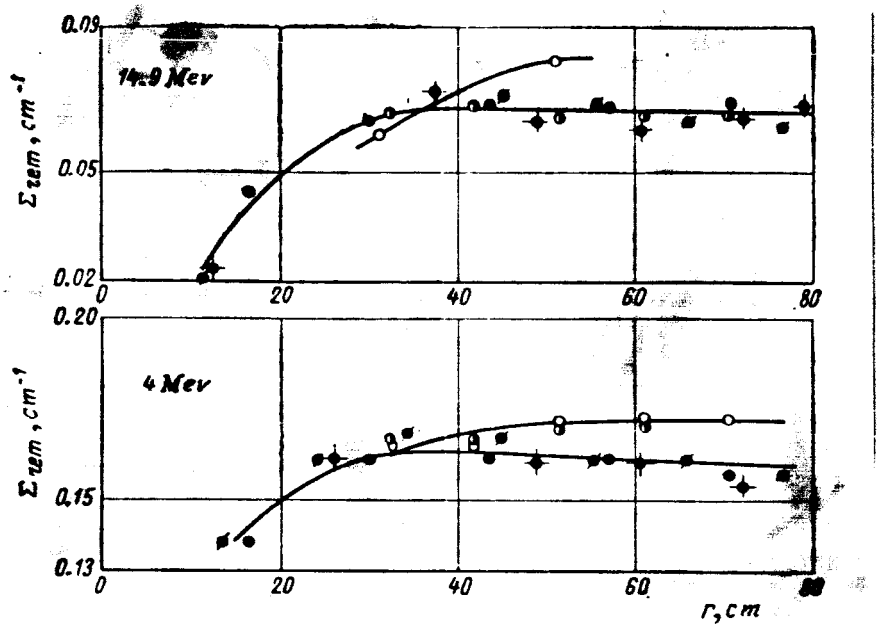

Fig.4 Dependence of the Removal Cross Section of Polyethylene and Plexiglas in Aluminum, for Neutrons of Energies E = 4 Mev and 14.9 Mev

o - Σ_{rem} of polyethylene obtained from measurements in heterogeneous designs (thickness of polyethylene block, 11.1 cm); ● - Σ_{rem} of plexiglas obtained from measurements in heterogeneous designs (thickness of plexiglas layers, 12.4 cm); and in homogeneous designs; ●′ - Thickness of plexiglas layers, 1 cm; -●- - Thickness of plexiglas layers, 2 cm; ● - Thickness of plexiglas layers, 4 cm.

assembled. The tanks with water, the prisms of polyethylene and plexiglas were 1000 mm in cross section. The measurements on the PW reactor were performed in barrier design. The geometry used in this experiment has been described in detail (Bibl.6). In these measurements, we analyzed the effects of γ-rays and slow neutrons on the count of the Th^{232} fission chamber. The effects were

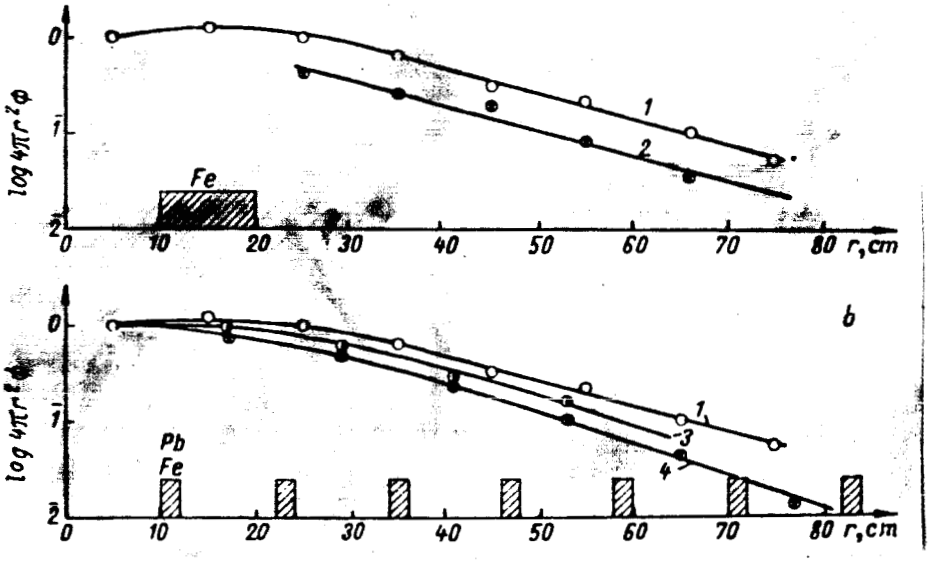

Fig.5 Spatial Distributions of Fast Neutrons from Neutron Source of Fission Spectrum in Boron Carbide behind an Iron Block (a) and in Homogeneous Mixtures of Boron Carbide with Lead and Iron (b)
1 - In boron carbide; 2 - In boron carbide behind iron block; 3 - In homogeneous mixture of boron carbide with lead; 4 - In homogeneous mixture of boron carbide with iron.

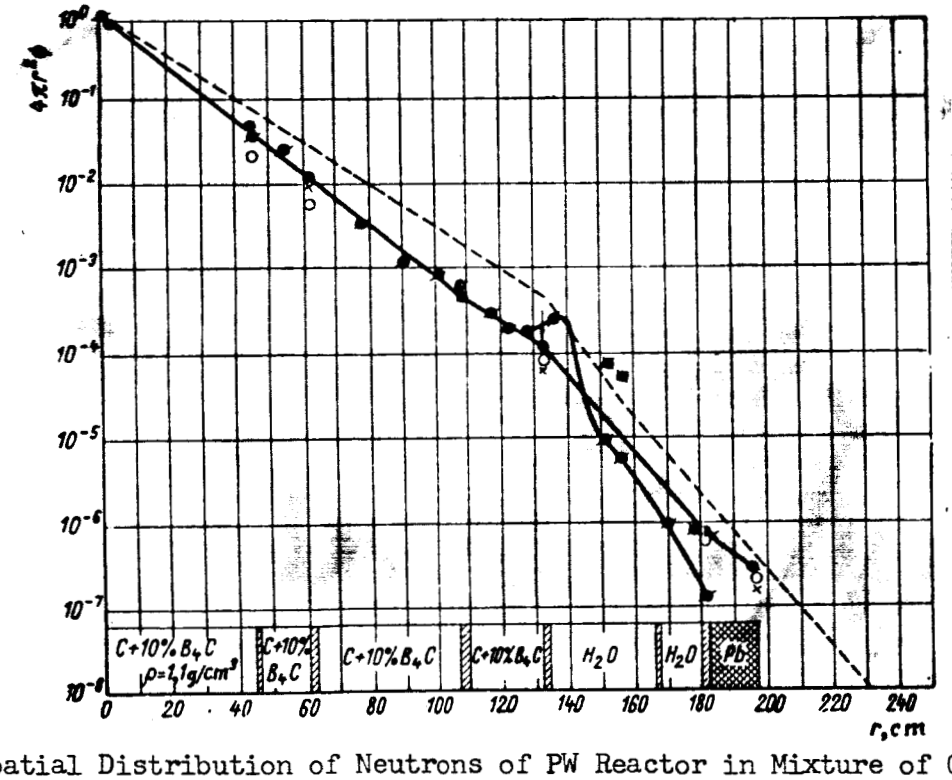

Fig.6 Spatial Distribution of Neutrons of PW Reactor in Mixture of Graphite with 7 wt.% of Boron (Infinite Design)
$\swarrow$ - U^{235} in cadmium; $\times$ - U^{235} without cadmium; × - ZnS (Ag); • - Th^{232} (fast neutrons); o - All-wave counter (total flux); - - - - - - - - Calculated (for carbon with boron carbide $\Sigma_{res} = 0.049\ cm^{-1}$); ——— - Experiment.

REMOVAL CROSS SECTIONS MEASURED IN WATER, BORON CARBIDE AND ALUMINUM

Material	Density g/cm³	Nuclear Density nuc/cm³ × ×10²⁴	$E_0 = 4$ Mev				E_0 14.9 Mev				Fission spectrum		
			Σ_{rem} in H_2O	Σ_{rem} in B_4C	Σ_{rem} in Al	$\Sigma = \frac{1}{\lambda_{rel}}$	Σ_{rem} in H_2O	Σ_{rem} in B_4C	Σ_{rem} in Al	$\Sigma = \frac{1}{\lambda_{rel}}$	Σ_{rem} in H_2O	Σ_{rem} in B_4C	Σ_{rem} in Borided graphite
Polyethylene CH_2	0.92	0.0394*	— —	— —	0.168 —	0.182[8] —	— —	— —	0.081 —	0.072 [8] —	0 112 [3] —	— —	— —
H_2O	1.00	0.0335*	0.159 [6,8] —	0.165 [4] —	— —	0 159 [6,8] —	0.076 [6,8] 0.066	0.084 [4] —	— —	[illegible] [6,8]	— —	— —	— —
Plexiglas $C_5H_8O_2$	1.18	0.00846*	— —	— —	0.171 0.161	0.160 [8] —	— —	— —	0.066 0.077	[illegible]	— —	— —	— —
Boron carbide B_4C	1.67	0.0182*	— —	0.083 [5] —	— —	0.083 [5] —	0.076 [9] —	0.058 [5] —	— —	0.058 [5] —	0.078 0.085	— —	— —
Graphite C	1.67	0.0836	— 0.094** [8]	— —	— —	0.088 [6] —	0.077 [9] 0.057** [8]	— —	— —	0.042 [6] —	0.068 [3] 0.075 [2]	— —	— —
Aluminum Al	2.7	0.0602	0.07 —	— —	0.071 —	0.071 —	0.087 [9] 0.066 —	— —	0.063 —	0.063 —	0.079 [1,3] —	— —	— —
Iron Fe	7.83	0.0848	— 0.160 [10]	0.169 [4] 0.171 [5]	0.162 0.162	0.132 [6,7]	0.134 [9] 0.113 [10]	0.137 [4] 0.136 [5]	0.145 0.145	0.121 [6,7]	0.167 [1,2] 0.168 [3]	0.1 0.1	0,154 —
Lead Pb	11.34	0.033	— —	0.113 [4] 0.113 [5]	0.101 0.092	0.067 [7] —	0.112 [9] —	0.097 [4] 0.096 [5]	0.084 0.088	0.0645 [6,7] —	[illegible] [2] 0.13 [1] 0.17 [3] —	— 0,112	— —

Note. For each substance, the upper horizontal row of values relates to measurements in heterogeneous block design and the lower, to measurements in homogeneous design. The literature source from which the removal cross section was taken is given in brackets.

* Nuclear density reduced to molecules per cubic centimeter.
** Obtained by calculation from the experimental attenuation curves in polyethylene and the calculated curves in hydrogen.

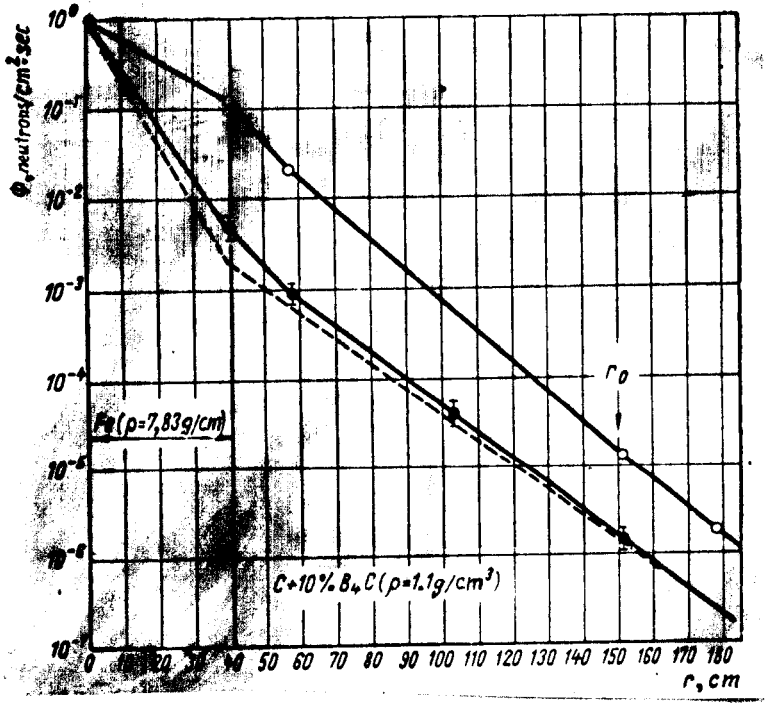

Fig.7 Spatial Neutron Distribution of the Spectrum of the PWR in a Mixture of Graphite with 7 wt.% Boron behind a Layer of Iron of 40 cm Thickness (Barrier Design)
– – – – – – Calculated (for iron Σ_{rem} = = 0.154 cm^{-1}); ———— Experiment; o - All-wave counter (total flux); ⊕ - Th^{232} (fast neutrons)

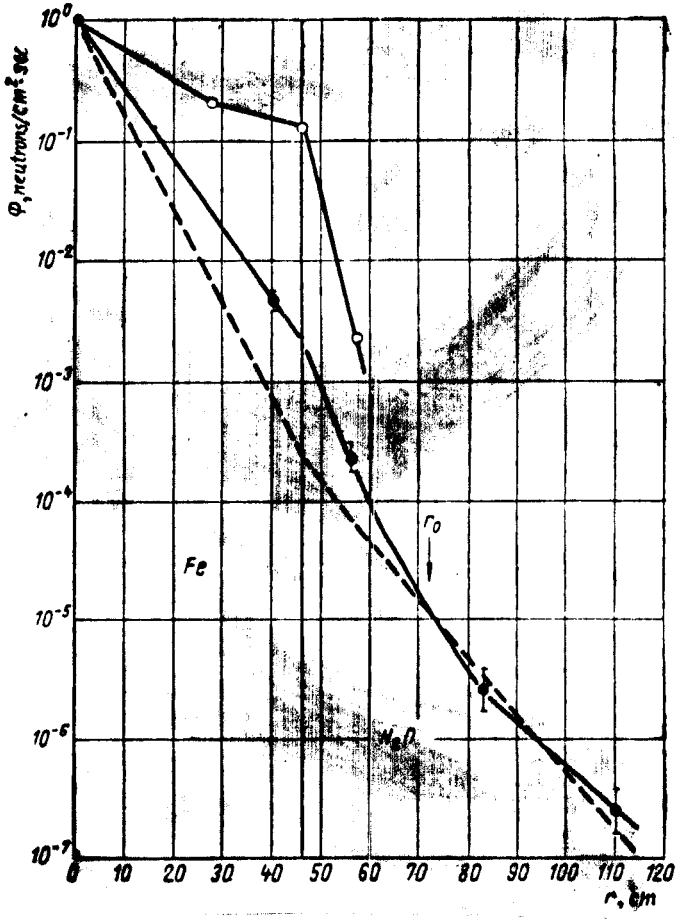

Fig.8 Spatial Neutron Distributions of the Spectrum of the PWR in Water behind a Layer of Iron of 46 cm Thickness (Barrier Design)
– – – – – Calculation (for iron Σ_{rem} = = 0.168 cm^{-1}); ———— Experiment; o - All-wave counter (total flux); ⊕ - Th^{232} (fast neutrons)

eliminated by the use of lead and boron carbide filters.

Measurements and Results

Figures 1 and 2 show the results of measurements in the Th^{232} fission chamber of the attenuation of the fast neutron fluxes in aluminum and its homogeneous mixture with iron and lead on neutron sources of E = 4 and 14.9 Mev. Similar measurements were made for assemblies of aluminum, in which blocks of iron, lead, plexiglas ($C_5H_8O_2$), and polyethylene (CH_2) were placed between the source and detector, and also for homogeneous mixtures of aluminum and plexiglas. Based on these measurements, we calculated the removal cross sections /59 for iron, lead, polyethylene, and plexiglas in aluminum (Figs.3, 4). The results of measurements on the spectrum of fission neutrons in boron carbide, in homogeneous mixtures of boron carbide with iron and lead and in boron carbide behind an iron block of 100 mm thickness are shown in Fig.5, while the removal cross sections of iron and lead, calculated from these data, are given in the accompanying Table.

The results of measurements in the PW reactor for graphite with an addition of 7 wt.% boron, and for borided graphite and water behind a layer of iron, are given in Figs.6 - 8. The dashed line shows the attenuation of the neutrons obtained by calculation, assuming a removal cross section of 0.168 cm for the measurements in water and of 0.154 cm for the measurements in borided graphite.

Discussion of the Results

The Table gives the removal cross sections for neutrons of the fission spectrum and monoenergetic neutrons with E = 4 and 14.9 Mev, measured in water, boron carbide, and aluminum. We have given both the removal cross sections

measured in our work and those obtained by other authors. The following conclusions can be drawn from a comparison of the values given in the Table and a consideration of the results presented in the diagrams.

1. The method of removal cross sections is applicable to calculations of shieldings in which light media other than water, for example boron carbide or aluminum, are used as moderator.

2. The value of the removal cross sections for most of the substances studied depends only negligibly on the choice of the moderator. Thus, on the spectrum of the PWR, the removal cross section of iron in borided graphite is 10% below that measured in water. This discrepancy, however, does not exceed the error of measurement.

It should be noted that, in principle, the method of removal cross sections can be used even in the case of very heavy moderators. However, the dependence of the removal cross section on the atomic number of the moderator, which is negligible at relatively low atomic weights of the moderator (at least up to A = 27), leads to a considerable underestimate of the removal cross section at moderators of very high atomic weight. Thus, the removal cross section of iron, measured in H_2O for neutrons of 4 Mev energy is 0.168 cm, while the same cross section, measured in an iron moderator (reciprocal of the relaxation length), is 0.132 cm. Furthermore, the removal cross section in a heavy moderator may depend largely on the energy threshold of the detector with which the cross section is measured.

3. In both homogeneous and heterogeneous reactors, the removal cross section attains saturation at relatively short distances r_0.

4. In boron carbide, as in water, measurements by a detector of the U^{235} chamber type give the same value for the cross sections as those measured by

means of threshold detectors of fast neutrons, i.e., Th^{232} chambers.

5. At a sufficient distance, greater than r_0, the removal cross section measured in a homogeneous mixture is close to that measured in a heterogeneous design.

6. The reciprocal of the relaxation length, at a sufficiently great distance from the source, is equivalent to the removal cross section of the given substance in the substance itself and for light media, as is obvious from the Table. This value can be expected as close to the removal cross section of the substance in other media. An exception is the value of λ_{rel}^{-1} for graphite at a neutron energy of 14.9 Mev. Apparently, such exceptions are possible in media with large "gaps" in the total cross section at energies lower than the initial neutron energy. In the cross section of carbon, such a gap is observed at an energy of 6.7 Mev. The removal cross section, measured in such media at great distances from the source, should correspond to the energy at which this gap occurs in the cross section of the moderator. This apparently explains the abnormally great relaxation length in graphite, for neutrons of an initial energy of 14.9 Mev. The reciprocal of the relaxation length in graphite for E = = 14.9 Mev is 0.0416 cm, while the removal cross section of graphite in water is 0.077 cm.

7. A practical application of the removal cross section method for /60
calculating the distribution of a group of fast neutrons with a depressed lower group boundary, using the values of the removal cross section obtained from measurements in water, is possible only in the presence of a light moderator. A typical example is the substantial difference between the values of λ_{rel}^{-1} and Σ_{rem} for iron and lead.

8. In the absence of a light component in the shielding, it is still pos-

sible to apply the method provided that the removal cross section of the material in the given medium is known or that the lower boundary of the energy group is significantly raised. Evidently, it is possible to select an energy boundary such that the removal cross section of the given material will not be dependent on the composition of the medium located behind the removing material.

The authors wish to express their thanks to V.P.Bogdanov, S.G.Osipov, G.V.Rykov, V.S.Tarasenko and A.I.Chusov who took part in the measurements.

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